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Part II

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Domestic Licensing of Production and Utilization Facilities; Interim Requirements Related to Hydrogen Control and Certain Degraded Core Considerations and Consideration of Degraded or Melted Cores in Safety Regulation

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Domestic Licensing of Production and Utilization Facilities; Interim Requirements Related to Hydrogen Control and Certain Degraded Core Consideratons

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The accident at Three Mile Island, Unit 2, resulted in a severely damaged or degraded reactor core with the concomitant release of radioactive material to the primary coolant system and generation of hydrogen from fuel cladding-water reaction well in excess of the amounts required to be assumed for design purposes by the current Commission regulations. Furthermore, the accident revealed design and operational limitations that existed relative to mitigating the consequences of the accident and determining the status of the facility during and following the accident. The Nuclear Regulatory Commission (NRC) is therefore initiating a long-term rulemaking to consider to what extent, if any, nuclear power plants should be designed to deal effectively with degraded-core and core-melt accidents. In the interim, the Commission is considering amending its regulations to improve hydrogen management in light-water reactor facilities and to provide specific design and other requirements to mitigate the consequences of accidents resulting in a degraded reactor core.

DATES: Comment period expires November 3, 1980. Comments received after November 3, 1980 will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments filed on or before November 3, 1980.

ADDRESS: Written comments or suggestions for consideration in connection with the proposed amendments should be submitted to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch. Copies of comments received may be examined in the Commission's Public Document Room at 1717 H Street NW., Washington, D.C.

FOR FURTHER INFORMATION CONTACT: Morton R. Fleishman, Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, telephone 301–443–5921.

SUPPLEMENTARY INFORMATION: The recent accident at Three Mile Island, Unit 2 (TMI–2), resulted in a severely damaged or degraded reactor core with a concomitant release of radioactive material to the primary coolant system and a large amount of fuel cladding metal-water (zirconium-oxygen) reaction in the core with hydrogen generation well in excess of the amounts required to be considered for design purposes by 10 CFR § 50.44, Standards for combustible gas control system in light water cooled power reactors. The accident revealed design and operational limitations that existed relative to mitigating the consequences of the accident and determining the status of the facility during and following the accident. To correct this situation the Commission has decided to revise its regulations so as to incorporate improvements derived from studies of the TMI–2 accident. The initial findings relative to the TMI–2 accident have been published in NUREG–0578, "TMI–2 Lessons Learned Task Force Status Report and Short-Term Recommendations," dated July 1979, and NUREG–0585, "TMI–2 Lessons Learned Task Force Final Report," dated October 1979.*

The NRC's Office of Nuclear Reactor Regulation has also sent letters to all licensees of operating nuclear power plants, operating license applicants, licensees of plants under construction and pending construction permit applicants, informing them of the followup actions that should be taken in light of the lessons learned from TMI–2. Specifically for all operating nuclear power plants, letters were sent on September 13, 1979 concerning "Followup Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident," in which a set of recommendations was presented that was to be implemented. Regional meetings were held during the week of September 24, 1979 to explain in more detail each of the recommendations. On October 30, 1979, letters were again sent to all licensees of operating nuclear power plants to provide additional clarification of the NRC staff recommendations (Discussion of Lessons Learned Short Term Requirements).

The Commission has identified a number of recommendations that are specifically related to accidents that involve severely damaged or degraded reactor cores. These have been determined to be of such safety significance that they should be codified by regulation in order to provide assurance that the public health and safety will be adequately protected. The specific dates to be required for implementation by the proposed amendments have been revised from the dates in the previously mentioned letters. However, the implementation dates are being reconsidered during this phase of the rulemaking and specific comments on this subject are welcomed.

The following discussion provides the background information, justification and clarification of the amendments, relative to degraded core accidents, that are proposed to be adopted by the Commission in response to the TMI–2 accident. Additional specific guidance for complying with the proposed amendments may be found in the previously mentioned letters to licensees of operating nuclear power plants. The proposed amendments themselves are presented after the discussion. The discussion is organized in the same order as the proposed amendments for ease in convenient cross reference.

Hydrogen Management (§ 50.44(c)(3) (i) and (ii))

Section 50.44, Standards for combustible gas control system in light water cooled power reactors, requires a licensee or license applicant to show that, during the time immediately following a postulated loss-of-coolant accident (LOCA) but before effective operation of the combustible gas control system, either: (1) an uncontrolled hydrogen-oxygen recombination will not take place in the containment, or (2) the plant can withstand the consequences of uncontrolled hydrogen-oxygen recombination without loss of safety. If neither of these conditions can be shown, the containment must be provided with an inerted atmosphere or an oxygen deficient condition in order to provide protection against hydrogen burning and explosions during this time.

Section 50.44 gives credit to performance of the emergency core cooling system (ECCS) by specifying that the amount of hydrogen assumed to be contributed by the metal-water reaction shall be either five times the total amount of hydrogen calculated in demonstrating compliance with the ECCS acceptance criteria (§ 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors) or an amount related to a specific depth of fuel clad reacted (approximately one percent of

*Copies of these reports may be obtained from GPO Sales Program, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

the fuel clad), whichever amount is greater. As a result, if a licensee or license applicant can show that the calculated metal-water reaction is well within the ECCS acceptance criteria (e.g., less than 0.5 percent of the fuel clad reacts which would be well within the 1.0 percent specified in the ECCs criteria), it is a straightforward task to demonstrate that there is no need to inert the containment of its plant, even if the plant has a small containment volume such as in a Mark I or Mark II boiling water reactor (BWR).

Most operating BWR plants with Mark I containments have been inerted due to guidance provided in an early version of Regulatory Guide 1.7, "Control of Combustibles as Concentrations in Containment following a Loss-of-Coolant Accident," in which the designs were required to accommodate five percent metal-water reaction during the LOCA blowdown. This guide was in effect prior to promulgation of § 50.44. The combination of containment volume and zirconium inventory of the BWR Mark I containments made it necessary to inert these containments to ensure that the hydrogen concentration in containment following a LOCA would not exceed the lower flammability limit of 4 volume percent in air. Section 50.44 and associated changes to Regulatory Guide 1.7 allowing credit for ECCS performance potentially reduced the amount of metal-water reaction that was required to be considered in the containment design. This permitted Hatch 2, a Mark I BWR plant located in Baxley, Georgia, and operated by the Georgia Power Co., to operate without inerting. In addition, it would permit Mark II BWR plants now in the operating license (OL) review process to operate without inerting. By a ruling of the Appeal Board (ALAB–229, September 18, 1974; 8 AEC 425 (1974)), inerting was not required for the Vermont Yankee plant, located in Vernon, Vermont, and operated by the Vermont Yankee Nuclear Power Corporation.

The Commission believes that pending completion of a long-term rulemaking proceeding related to consideration of degraded or melted cores in safety regulation, which will include a reevaluation of all positions on the inerting of reactor containments, it would be prudent, in the interim, to establish certain inerting requirements for operating plants and those that will be operated in the near term as follows: (1) Mark I and II BWR containments be inerted. While the decrease in residual risk due to inerting these containments is small, as determined by probabilistic analyses, (a) there are no significant countervailing safety disincentives; (b) the cost of inerting is small; and (c) there has been substantial satisfactory experience with inerting Mark I containments. (2) Ice condenser containments for pressurized water reactor (PWR) plants not be inerted at this time. While inerting ice condenser containments would result in a greater decrease in the calculated residual risk, compared to BWR containments, there are significant reasons for not inerting such as, (a) inerting would result in an increased personnel hazard, since these containments must be entered frequently during routine operations for maintenance purposes; (b) inerting could lead to a decrease in safety performance of the ice consenser because of increased difficulty in maintenance; and (c) inerting is not within proven technology for this type of containment and would have to be thoroughly analyzed and tested (with likely changes in design) before imposing such a requirement. (3) Other PWR containments not be inerted at this time. While inerting would likely produce a small decrease in residual risk, any decrease would be substantially less than for Mark I or II BWR containments because of the substantially larger containment volumes and the fact that these containments can probably withstand higher pressures from hydrogen burns. Both of these considerations would permit substantially greater volumes of hydrogen to be generated.

The Commission is therefore considering amending § 50.44 to require that: (1) all operating Mark BWRs be inerted; (2) Mark I and II BWRs now under construction be inerted upon operation; and (3) licensees now operating PWR plants or Mark III BWR plants, or license applicants that plan to operate these plants, study the various methods of controlling the behavior of large amounts of hydrogen before the containment is threatened, or of mitigating the consequences of accidents involving the generation of large amounts of hydrogen. These studies should take the adverse safety aspects of the design features as well as their costs into account, recognizing that tradeoffs may be necessary. They should cover a range that includes both the realistic (best estimate) ultimate strength of the containment as well as the conservative design basis. The maximum duration for hydrogen generation was specified to be eight hours because it was felt this provided an adequate limit based on credible accident scenarois. It neglects the hydrogen generated by radiolytic decomposition which is more than adequately accounted for by the assumption used for the percent of fuel clad that reacts. For purposes of providing a reasonable bound to the studies, an upper limit of 75 percent should be used for the percent of fuel clad that reacts to produce hydrogen. As indicated previously, this can also account for the hydrogen that would be generated from other possible sources such as from the stainless steelsteam reaction or radiolytic decomposition. The Commission will require that the studies include sensitivity analyses concerning the effects of different amounts of hydrogen, and that the methods to be considered for hydrogen control include but not be limited to: (1) inerting of containment; (2) use of hydrogen recombiners; (3) use of purge systems; (4) use of a halon suppressant system; (5) use of a filtered-vent system; (6) use of a hydrogen combustion system; (7) use of water fog spray; and (8) combinations of these. It is anticipated that, where appropriate, owners groups will be formed for similar facilities to perform these studies and thus eliminate duplication.

It should be noted that this proposed amendment will require that the Mark I containments for the Vermont Yankee and Hatch 2 BWR plants be treated in a manner similar to other operating BWR plants with Mark I containments. The first domestic BWR Mark II is scheduled for fuel loading in late-1980, and the first domestic BWR Mark III is scheduled for fuel loading about September 1981.

Additional information and discussion on the inerting of light-water-reactor containments may be found in SECY–80–107, "Proposed Interim Hydrogen Control Requirements for Small Containments," and SECY–80–107 A and B, "Additional Information Re: Proposed Interim Hydrogen Control Requirements,"* and NUREG–0578 and NUREG–0585 which were referenced earlier.

The Commission realizes that amending § 50.44 to require inerting of Mark I and II BWRs does not settle the issue of how to treat degraded core accidents in general and hydrogen management in particular. There are numerous regulations which may be affected by the consideration of degraded core accidents. Some of these are Criteria 16 and 50 of Appendix A to 10 CFR Part 50, which are related to containment design, and § 100.11 of 10

*Copies of those documents are available from the Commission's Public Document Room at 1717 H Street NW., Washington, D.C.

CFR Part 100, which is related to the siting of nuclear reactors. Also, the remainder of § 50.44, particularly in regard to the hydrogen generation assumptions, is not being considered for revision at this time. Since § 50.44(d) is not being changed, the hydrogen control system does not have to accommodate any more hydrogen than that associated with either five times the metal-water reaction calculated to occur during a postulated loss-of-coolant accident or a one percent metal-water reaction, whichever is greater. The Commission is performing a systematic review of all of its regulations from the standpoint of consistency relative to the treatment of degraded core accidents. Furthermore, the Commission has initiated a longer term rulemaking effort relative to consideration of degraded or melted cores in safety regulation that would identify, among other things, whether and to what extent designers need to consider degraded core accidents.

Dedicated Hydrogen Control Penetrations (§ 50.44(c)(3)(iii))

Paragraphs 50.44(d), (e), (f), and (g) of 10 CFR Part 50 require the incorporation of hydrogen recombiners or post-accident purge systems for the control of combustible gas concentrations inside containment, depending on the date of the notice of hearing on the application for the construction permit (CP) for the plant. All plants for which the CP hearing notice was published after November 5, 1970, must use combustible gas control systems such as hydrogen recombiners for the post-accident control of combustible gas in the containment building. Plants for which the notice of hearing was published prior to November 5, 1970, may use post-accident purging of the containment for combustible gas control depending on whether or not they meet certain site radiation dose criteria.

The TMI–2 plant had provisions for post-accident installation and operation of an external hydrogen recombiner for combustible gas control. However, the design of the external recombiner hookup at TMI–2 use the 36-inch containment penetrations for the normal containment purge system by tapping 4-inch lines off the purge lines outside the containment building between the building and the outer containment isolation valves. Operation of the hydrogen recombiner required the opening of the inboard 36-inch containment isolation valve in both a containment purge system inlet and outlet line. With this design, once the hydrogen recombiner is put into operation, containment integrity is vulnerable to a single active failure. That is, a spurious or inadvertent opening of one of the 36-inch outboard containment isolation valves would result in the venting of the containment to the environment.

The purpose of this proposed amendment is to provide assurance that facilities with hookups for external recombiners or post-accident purge systems that are susceptible to single failures, and that would result in operation of the unit beyond its design capacity, have design modifications made to correct this situation. Systems designed to meet these proposed requirements would not provide through-line leakage paths between the containment atmosphere and the environment and would eliminate the possibility of violating the containment integrity through a single active failure during hydrogen recombiner or purge system operation.

Hydrogen Recombiner Capacility (§ 50.44(c)(3)(iv))

As discussed previously, existing regulations permit plants for which the notice of CP hearing was published prior to November 5, 1970 to use only purge systems for combustible gas control subject to certain site radiation dose criteria. Depending on the containment design for these plants, the design basis for the purge system would lead to a release of the containment atmosphere for long-term hydrogen control in a period of time varying from a few weeks to several months.

Recombiners designed to meet current Commission requirements are not capable of preventing the hydrogen combustion and the resultant containment pressure spike that occurred in the TMI–2 containment building during the course of the accident. However, had there not been the capability to use a recombiner at TMI–2, it is possible that under a different accident scenario venting of the highly radioactive containment atmosphere may have been necessary in the weeks following the start of the accident.

As indicated in the discussion under Hydrogen Management, the Commission has initiated a long-term rulemaking effort relative to consideration of degraded or melted cores in safety regulation. Part of this rulemaking will involve a thorough reevaluation of hydrogen generation and control including the design bases for current hydrogen recombiners. However, based on the TMI–2 experience, and the desire to reduce the likelihood of releasing significant amounts of radioactive material to the environment, the Commission believes that it would be prudent to provide means for hydrogen control that do not involve venting to the atmosphere. Because recombiner technology is well established and relatively inexpensive, the Commission is considering amending its regulations (new § 50.44(c)(3)(iv)) to require that all plants that currently rely on venting as the primary means for combustible gas control be provided with the capability to install external hydrogen recombiners following the start of an accident. This will require the use of dedicated containment penetrations as well as appropriate shielding, electrical power and operational procedures. The recombiner capability would only be required to satisfy the combustible gas control requirements of § 50.44 and thus would not be required to have any more capability than that needed to control hydrogen resulting from at most a 5 percent metal water reaction. This would provide a long-term hydrogen control capability but not a capability to quickly reduce the concentration of large amounts of hydrogen such as was generated during the TMI–2 accident. The need for this capability will be addressed during the long-term rulemaking on consideration of degraded or melted cores in safety regulation.

High Point Vents in the Reactor Coolant System (§ 50.44a(a))

During the TMI–2 accident, a substantial volume of hydrogen was generated in the primary system. Helium fill-gas and fission product noble gases were also released. In other reators under different circumstances, nitrogen, which drives the passive injection water system in PWRs, may also be a potential source of gas in the primary system. There is a concern that the accumulation of pockets of noncondensible gases in the primary system of a reactor may interfere with the natural circulation pattern that is regarded as an important safety feature in some accident sequences. In other sequences, pockets of noncondensible gases may interfere with pump operation. It has therefore been concluded that under certain circumstances it would be desirable to have provisions for venting noncondensible gases from high points in the primary system. The vents would have to be operable from a remote location, such as the control room.

Furthermore, the introduction of hydrogen from the primary system into the containment building can create a flammable atmosphere unless precautions are taken against this eventuality. Obviously, the decision to place such a vent system into operation

would have to be made with the greatest care and restraint.

The purpose of this proposed amendment is: (1) to provide reactor coolant system and reactor vessel head high point vents remotely operable from the control room in reactors where these are not already in place; and (2) to provide assurance that the use of these vents will not further aggravate the challenge to containment or the course of the emergency. This proposed requirement is also interpreted to mean that other systems which may be required to maintain adequate core cooling such as the isolation condenser in BWRs, and the decay heat removal system, have the capability to be remotely vented if the accumulation of noncondensible gases would cause their loss of function. It is recognized that it is impracticable to vent each of the many thousands of tubes in a U-tube steam generator. Procedures should be developed which ensure that sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed from the reactor coolant system. Proposed § 50.44a(g), "Training to Mitigate Degraded Core Accidents," includes plans and training for the use of the high point vents in emergency situations.

Protection of Safety Equipment and Areas Which May Be Used During and Following and Accident (§ 50.44a(b)(1))

During and following an accident that results in a severly degraded core, a large amount of radioactive material may be relased from the fuel as a result of cladding failure and be carried throughout the facility by the water in which it is dispersed, and by the circulating air. Systems that were not designed to contain large amounts of radioactive material, such as the auxiliary building at TMI–2, may become highly contaminated. The resulting radiation fields may make it difficult to effectively perform accident recovery operations or may impair the functioning of systems important to safety. These systems, although not specifically identified to perform post-accident functions, may nevertheless be of significant value after an accident. In addition, areas which may be used during and following an accident, such as control rooms, radwaste panels, emergency power supplies, and instrument areas may fall within the radiation fields of these systems.

Procedures for the use of these areas during and following an accident may be all that is necessary. In other instances, permanent or temporary shielding may be valuable. Remote instrument and control capability may also solve some problems. Finally, the solution may require basic design improvements in the facilities, components or systems.

Areas that may be used during and following an accident should be provided with appropriate shielding and other radiation protection measures so that personnel performing necessary safety functions will not receive more than a 5 rem whole body dose or its equivalent to any part of the body. When determining anticipated personnel doses for the purpose of meeting this facility design criterion, care must be taken to determine the necessary occupancy time in a specific area. For example, areas in which there will be continuous occupancy are required to have much lower design dose rates than areas where minimal occupancy is expected. Therefore, the facility design should be based upon expected occupancy, as well as the radioactivity present and the shielding available.

The purpose of the proposed design requirement (§ 50.44a(b)(1)) is to facilitate operations during and following an accident in areas affected by systems that may contain abnormally high levels of radiocativity and to ensure that safety equipment in proximity to the resulting radiation fields is not unduly degraded. It should be noted that the radioactive material release specified in § 50.44a(b)(1)(i) represents the Commission's position at this time; however, this position will be reevaluated during the long-term rulemaking on consideration of degraded or melted cores in safety regulation.

In-Plant Iodine Instrumentation (§ 50.44a(b)(2))

10 CFR Part 20 provides criteria for control of exposures of individuals to radiation in restricted areas, including airborne radioiodine. Since iodine concentates in the thyroid gland, airborne concentrations must be known in order to evalaute the potential dose to the thyroid. If the airborne radioiodine concentration is overestimated, plant personnel may be needlessly required to perform operations while wearing respiratory protective equipment. Such action can sharply limit communication capability and may diminish personnel performance during an accident.

The concentration of radioiodine in air is determined by use of an iodine adsorbing cartridge through which air has been pumped. The cartridge is removed from the air pump and allowed to ventilate to permit any adsorbed noble gases to difuse to the atmosphere. The cartridge is then analyzed for radioactivity content, which is ascribed to radioiodine. While this procedure is conservative, it is possible for sufficient noble gas to be adsorbed and not released from the charcoal that the resulting radioiodine determination may be unduly conservative (high). Because the radioiodine concentration was greatly overestimated at TMI–2, plant personnel were required to perform their operations while wearing respiratory protective equipment when this use was not necessary. Actual concentrations were probably below levels requiring protective actions. One acceptable method to minimize this problem is to measure the radioiodine by gamma energy spectrum analysis. Equipment for these measurements is commercially available.

Effective monitoring of radioiodine levels in the building, under accident conditions, could include the use of portable analyzers and instruments. Alternatively, the capability should be provided to remove the filter sample to a low background, low contamination area, for analysis, to prevent measurement inaccuracies due to background radiation.

The purpose of the proposed requirement (§ 50.44a(b)(2)) is to improve the accuracy of measurement of airborne radioiodine concentrations within nuclear power plants.

Sampling During and Following an Accident (§ 50.44a(c))

Timely information from reactor coolant and containment air samples can be important to reactor operators for their assessment of system conditions can influence their subsequent actions to maintain the facility in a safe condition. During and following an accident, significant amounts of fission products may be present in the reactor coolant and containment air, creating high radiation levels throughout the facility. These high radiation levels may delay the obtaining of information from samples. The high background radiation, high sample radiation, and high levels of airborne contamination may render in-plant radiological spectrum analysis equipment inoperable during and after an accident.

At TMI–2, all of the above problems were encountered. There was no capability to obtain and analyze in a timely manner the reactor coolant and containment air samples under the existing accident conditions. The acquisition of reactor coolant and containment air samples was delayed for several days while personnel radiation protection precautions were taken. Once the samples were obtained,

there were significant delays in performing the radiological spectrum analysis of the samples. The TMI–2 spectrum analysis equipment was inoperable because of high background radiation and the samples had to be packaged and flown to an offsite laboratory for analysis.

In summary, radiation levels at TMI–2 delayed acquisition of information to confirm that significant core damage had occurred. Prompt acquisition and analysis of reactor coolant samples within a few hours after the initial scram would have indicated that significant core damage had occurred. With such information, earlier remedial actions could have been taken. Similarly, analysis of an early containment air sample would have indicated the presence of hydrogen, and of radioisotopes indicating significant core damage, and the potential for a hydrogen explosion in the containment.

The sampling facilities are required to meet the design criteria of § 50.44a(c) as well as those of § 50.44a(b)(1). All sources of radiation should be considered in evaluating the designs, such as the sample lines, the samples themselves, and other radioactive lines near the sampling stations and analysis facilities.

Leakage Integrity Outside Containment (§ 50.44a(d))

Several of the engineered safety features and auxiliary systems, located outside reactor containment, will or may have to function during a serious transient or accident with large radioactive inventories in the process fluids. The leakage from these systems, when operated, must be minimized or eliminated to prevent the release of significant amounts of radioactive materials to the environment. Examples of engineered safety features include residual heat removal, containment spray recirculation, and high-pressure injection recirculation. Examples of auxiliary systems include sampling, makeup and letdown, and waste gas. These systems are checked during pre-operational testing and startup testing but are not usually included in any periodic leak testing program. It is important that the plant operating staff know the leakage rates of these systems and minimize this leakage to the maximum extent practicable.

Some of these systems were used during the TMI–2 accident with resultant releases of radioactive materials to the auxiliary building ventiliation systems. These releases are believed to have resulted from leaking relief valves, waste gas compressor seals, valves, manifold lines, and open rupture discs. Uncertainties in the leakage rate of the residual heat removal system could have hindered its use if it had been needed. The lesson learned in this case was that more positive control and knowledge of the leakage rates of these systems is needed to provide the operating staff with the maximum usable equipment and to restrict or control the release of radioactive materials to the environment.

The purposes of this proposed amendment are to require that every reasonable effort be made to eliminate or reduce the leakage from these systems, to require that periodic tests are performed to ensure that the leakage from these systems is minimized to the maximum extent practicable, and to provide the plant staff with current knowledge of the system leakage rates.

Accident Monitoring Instrumentation (§ 50.44a(e))

The accident at TMI–2 demonstrated that containment conditions can arise that are more severe than those that were postulated to occur during design basis accidents.

Approximately ten hours after the start of the accident at TMI–2, a 28-psig pressure spike occurred in the containment building. It is believed that the pressure spike was due to the rapid burning of hydrogen gas in the containment atmosphere. It is known that the pressure spike represented a serious condition within containment and that the pressure indication itself could have been, but was not then accepted as, critical information to the plant operators. The events at TMI–2 clearly reaffirm the need for containment pressure indication in the control room. The instrumentation range should extend from a lower limit of five psi below normal operational pressure to an upper limit of three times the design pressure for concrete containments and four times the design pressure for steel containments.

The sequence of events during the accident at TMI–2 indicate another item of information which could have been, but was not immediately accepted as, critical information in the diagnosis of the accident, namely the free liquid inventory in the containment building. During the accident, reactor primary coolant water vented through the drain tank relief valve and drained to the containment sump. Prior to containment isolation, water in the containment sump was discharged to the auxiliary building sump tank. Because containment sump pump operation had routinely occurred several times a day before the accident, the transfer process was not recognized as a potential source of contaminated water in containment. Additionally, the accumulation of water in the TMI–2 containment probably contributed to equipment failure due to flooding. The events clearly establish a need for accurate containment water level indication in the control room, with instrument ranges which include maximum accident flooding levels.

The third item of information which was subsequently considered to be of critical importance in determining containment conditions at TMI–2 was a measurement of the hydrogen concentration in the containment atmosphere. Hydrogen gas was produced as a result of the reaction of zirconium metal from the fuel cladding and primary coolant water in the reactor core. The gas was vented from the reactor coolant system to the containment atmosphere. The free hydrogen in containment resulted in a rapid burn and pressure spike event in the containment. Samples of containment atmosphere were taken following the accident at TMI–2, but the process involved risk to workers and did not yield real-time information. The events clearly show that it is essential that the operator have continuous information about the containment atmosphere hydrogen concentration for indication of the need and use of reactor pressure vessel venting or containment combustible gas control systems.

Indication of containment pressure, containment water level, and hydrogen concentration in the containment atmosphere will provide critical information to the operator on containment conditions during and following an accident and should be qualified to the criteria for instrumentation important to safety. These parameters should be continuously provided and recorded in the control room of all nuclear reactor power plants.

The radiation level inside containment is closely related to the potential release of radioactive materials in the plant effluents. At TMI–2, the radiation monitor in containment had a range limit of 10^6 R/hr, which was adequate to meet the conditions of the accident. In reviewing the monitoring capabilities of other plants, however, it was found that there are few operating plants with instrumentation capable of measuring levels in excess of 10^6 R/hr. During the initial post-accident period at TMI–2, questions arose as to the validity of the instrument readout and to the operational characteristics of the instrument under the accident environment. The Commission considers that the high-level monitoring

instrumentation in the containment at TMI–2 was adequate to measure the existing radiation levels; however, it also considers that the range limit of this instrumentation should be increased to greater than 10^6 R/hr and consit of at least two channels, each separated physically from the other, and that the instrumentation system should be qualified to the criteria for instrumentation important to safety.

Guidance on the ranges and specifications of the accident monitoring instruments discussed above can be found in proposed Revision 2 of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,"* which is presently out of public comment.

Radioactive gaseous effluent monitors designed to operate under conditions of normal operation and anticipated operational occurrences do not have sufficient range to function under release conditions associated with certain types of accidents. General Design Criterion 64 of Appendix A to 10 CFR Part 50 requires that effluent discharge paths be monitored for radioactivity that may be released from postulated accidents. The gaseous effluent monitoring system for TMI was evaluated during the licensing review and was found to be adequate for calculated releases from previously postulated accidents; however, the TMI–2 experience gives rise to new postulated design basis accidents and their associated releases.

At TMI–2, the noble gas section of the gaseous radioactive effluent monitor serving the plant vent was designed to measure concentrations up to 10^{-2} μCi/cc (Xe–133). During the initial phases of the accident, radioactive noble gas effluent monitor readings were off scale, with actual release concentrations calculated to be on the order of 10^{-1} μCi/cc to 1 μCi/cc.

A section of the TMI–2 plant vent gaseous radioactive effluent monitor designed to detect and measure radioiodine releases, while remaining on scale, gave an erroneous indication of high radioiodine content in releases from the vent during the initial phases of the accident. The indication was caused by adsorption of short-lived noble gases in the charcoal cartridge, with the presence of the noble gases being read and erroneously interpreted as radioiodine by the monitor readout system.

A similar condition existed in the section of the plant vent monitor designed to detect and measure the presence of particulate radioactive material in plant gaseous effluents. In this case, the presence of noble gases in the gas stream passing through the monitor's particulate filter was sufficient to cause the particulate section of the monitor to read off scale and erroneously indicate that large quantities of particulates were being released from the plant vent.

The problem is generic. A recent survey of existing radioactive gaseous effluent monitoring capabilities of operating plants shows that less than 20 percent of operating plants have monitors that would have stayed on scale under the conditions of the TMI–2 accident. Also, it can be shown that the potential releases from postulated accidents may be several orders of magnitude higher than was encountered at TMI–2. Under such circumstances, none of the effluent monitors now in service at any operating plant would remain on scale.

A requirement for effluent monitors to have an operating range sufficient to permit on-scale readings under accident conditions is needed to provide meaningful release information for offsite emergency actions. Examples of anticipated release points that should be monitored are: the auxiliary building, radwaste building, waste gas decay tanks, main condenser air ejector, BWR main condenser vacuum pump exhaust, PWR steam safety valves and atmosphere steam dump valves, and BWR turbine building. Other examples would be areas that communicate directly with systems which may contain primary coolant or containment gases, such as letdown and emergency core cooling systems, and external hydrogen recombiners.

Based on data submitted by plant operators, the installed capability exists for monitoring noble gas releases up to a concentration of approximately 1×10^3 μCi/cc, which is a factor of 10^5 higher than the maximum range of the instrumentation in use at TMI–2.

It has been calculated that the maximum concentration of undiluted noble gases which could be present in effluent gaseous releases is on the order of 10^5 μCi/cc. The Commission considers the upper detection limit of 10^5 μCi/cc for noble gases to be technically achievable. Guidance on the ranges and specifications for the required instruments can be found in proposed Revision 2 of Regulatory Guide 1.97.

The Commission understands that technological problems exist in monitoring of radioactive particulates and radioiodines in accident level releases, at calculated maximum radioiodine concentrations on the order of 10^2 μCi/cc. It has therefore concluded that sampling of plant gaseous effluents, with laboratory analysis of samples subsequent to release, is the only currently valid technique for monitoring accident level releases of radioiodines and particulates. The proposed amendment would require that this technique be used until on-line monitoring capability for accident level releases is developed. Existing radioiodine effluent monitoring equipment should be used to monitor radioiodine releases within instrumentation design limitations, with sampling and analysis being used to extend monitoring capabilities to accident levels.

Detection of Inadequate Core Cooling (§ 50.44a(f))

During the TMI–2 accident, the condition of low water level in the reactor vessel and inadequate core cooling was not recognized for a long period of time. This problem was the result of a combination of factors including an insufficient range of existing instrumentation, inadequate emergency procedures, inadequate operator training, unfavorable instrument location (scattered information), and perhaps insufficient instrumentation. The instrumentation already available on some operating reactors that could indicate inadequate core cooling includes core exit thermocouples, cold leg and hot leg resistance temperature detectors (RTDs), in-core neutron detectors, ex-core neutron detectors, and reactor coolant pump current meters. Generally, such systems were included in the reactor design to perform functions other than monitoring of core cooling or indication of vessel water level. The as designed and field-modified instrumentation at TMI–2 did not provide sufficient information to indicate reduced reactor vessel coolant level, core voiding, and deteriorated core thermal conditions to the reactor operators at the time of the accident, although, in retrospect, this could be inferred from the instrument records. An additional diagnostic tool that was not available was a continuous indication of the margin to saturated conditions in the primary coolant. A meter to provide this information is proposed to be required for all PWRs.

The purpose of this proposed amendment is to provide the reactor

*Copies of this draft regulatory guide may be obtained from the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

operator with instrumentation, procedures, and training necessary to readily recognize and implement actions to correct or avoid conditions of inadequate core cooling.

Training to Mitigate Degraded Core Accidents (§ 50.44a(g))

The TMI–2 accident has pointed out the need to improve the capability to recognize, diagnose, control and mitigate the consequences of accidents resulting in a degraded reactor core. That is, the operating personnel should have the capability to respond to events that are more severe than those transients and accidents considered as the design basis for the plant.

The purpose of this proposed amendment is to require additional training, for all operating personnel, specifically in relation to accidents involving a degraded core. This training will prepare the operating personnel to use all the available instrumentation and equipment to properly respond to such accidents.

In addition to the proposed interim rulemaking, the Commission has initiated a long-term rulemaking in which it is considering whether and to what extent its regulations should be amended to deal effectively with a broad range of accidents involving a reactor whose core has been degraded or melted. As part of the long-term rulemaking, an advance notice of proposed rulemaking has been issued soliciting comments on certain specific questions. Any views and comments submitted in response to this notice of proposed interim rulemaking will also be considered, along with comments submitted in response to the advance notice of proposed rulemaking, in the overall long-term rulemaking related to consideration of degraded or melted cores in safety regulation. Pending completion of the long-term rulemaking, it is the Commission's judgment that implementation of the requirements hereby being proposed will, in the interim, provide sufficient assurance that the health and safety of the public will be adequately protected relative to degraded core accidents. Other documents which support this position are SECY–80–107, 107A, 107B, NUREG–0578 and NUREG–0585 which were referenced previously.

(Separate views of Commissioners Gilinsky and Bradford are attached.)

Accordingly, notice is hereby given that, pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and section 553 of title 5 of the United States Code, adoption of the following amendments to 10 CFR Part 50 are contemplated.

PART 50—DOMESTIC LICENSING OR PRODUCTION AND UTILIZATION FACILITIES

1. Section 50.44 of 10 CFR Part 50 is amended by revising paragraph (c) to read as follows:

§ 50.44 Standards for combustible gas control system in light water cooled power reactors.

* * * * *

(c)(1) For each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding, it shall be shown that during the time period following a postulated LOCA but prior to effective operation of the combustible gas control system, either: (i) An uncontrolled hydrogen-oxygen recombination would not take place in the containment; or (ii) the plant could withstand the consequences of uncontrolled hydrogen-oxygen recombination without loss of safety function.

(2) If neither of these conditions can be shown, the containment shall be provided with an inerted atmosphere or an oxygen deficient condition in order to provide protection against hydrogen burning and explosions during this time period.

(3) However:

(i) As soon as practicable but not later than June 30, 1981, an inerted atmosphere shall be provided for each boiling light-water reactor facility for which the application for a construction permit was docketed between March 15, 1964 and July 1, 1972; and

(ii) Design analyses shall be performed (A) for each boiling light-water reactor facility for which the application for a construction permit was docketed after July 1, 1972 and, (B) for each pressurized light-water reactor facility, to evaluate measures that can be taken to mitigate the consequences of large amounts of hydrogen generated within 8 hours after the start of an accident (hydrogen resulting from the reaction of up to about 75 percent of the fuel cladding with water). These design analyses and a proposed design (or designs) to mitigate the consequences of hydrogen in containment shall be completed and submitted to the Commission by (6 months from effective date of rule) or the date of docketing of the application for the operating license, whichever is later.

(iii) By June 30, 1981, facilities that rely upon external recombiners or purge systems to satisfy the requirements of § 50.44 shall be provided with containment penetrations for the external recombiners or purge systems that either:

(A) are dedicated to that service only, conform to the requirements of Criteria 54 and 56 of Appendix A, are designed against an assumed single failure, and are sized to satisfy the flow requirements of the external recombiners or purge systems, or

(B) are of a combined design for use by either external recombiners or purge systems and other systems, conform to the requirements of Criteria 54 and 56 of Appendix A, are designed against an assumed single failure both for containment isolation purposes and for operation of the external recombiners or purge systems, and are sized to satisfy the flow requirements of the external recombiners or purge systems.

(iv) By January 1, 1982, facilities that rely upon purge systems as the primary means for controlling combustible gases following a LOCA shall be provided with the capability to install external recombiners following the start of an accident that meet the combustible gas control requirements of this section. The containment penetrations that are used must meet the criteria in paragraphs (c)(3)(iii)(A) and (c)(3)(iii)(B) of this section applicable to external recombiners.

* * * * *

2. A new § 50.44a is added to 10 CFR Part 50 to read as follows:

§ 50.44a Design requirements to mitigate the consequences of accidents resulting in a degraded core.

(a) *High Point Vents.* By January 1, 1982, each boiling and pressurized light-water nuclear power reactor shall be provided with reactor coolant system and reactor vessel head high point vents [1] remotely operated from the control room to provide improved operational capability for maintaining adequate core cooling following an accident. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents and associated controls, instruments and power sources must conform to the requirements of Appendix A and Appendix B. In particular, these vents shall be designed in such a way that no single failure could result in either (1) a loss of the capability of the vents to perform their safety functions or (2) inadvertent actuation of a vent.

(b)(1) *Protection of Safety Equipment and Areas Which May be Used During and Following an Accident.* By January 1, 1982, each boiling and pressurized light-water nuclear power reactor shall

[1] Except that the tubes in U-tube steam generators do not require venting.

be provided with both adequate access to areas which may be used during and following an accident and protection of safety equipment so that an accident, which results in the release of large amounts of radioactive material, will not limit personnel occupancy or degrade safety equipment, by the radiation fields that may exist during and following the accident, to the extent that required safety functions cannot be accomplished.

(i) The facility design must be based on a release of radioactive material from the fuel to the primary coolant system that is not less than 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory, and 1% of the remaining core fission products. For equipment and areas affected by the ractor coolant, it shall be assumed that the above distribution of radioactive material is intimately mixed with the coolant water. For equipment and areas affected by the containment atmosphere, it shall be assumed that not less than 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen inventory are uniformly dispersed in the containment atmosphere and an additional 25% of the core equilibrium halogen inventory and 1% of the remaining core fission products are uniformly distributed on surfaces exposed to the containment atmosphere.

(ii) The facility design must be such that an individual operator will not receive more than a 5 rem whole body dose, or its equivalent to any part of the body, while performing a necessary safety function during and following an accident.

(2) *In-Plant Iodine Instrumentation.* By January 1, 1981,[2] each boiling and pressurized light-water nuclear power reactor shall be provided with instrumentation, equipment and associated training and procedures for determining, under accident conditions, the airborne radioiodine concentration in areas within the facility where plant personnel may be present during and following an accident.

(c) *Sampling During and Following an Accident.* By January 1, 1982, each boiling and pressurized light-water nuclear power reactor shall be provided with the capability for personnel to obtain and quantitatively analyze a reactor coolant or containment atmosphere sample during and following an accident.

(1) The facility design must be based on the radioactive material release terms described in paragraph (b)(1)(i) of this section.

(2) The capability to obtain a sample must include the capability for doing so, promptly, and without incurring a radiation exposure to any individual in excess of 3 rem to the whole body, or its equivalent to any part of the body.

(3) The capability to quantitatively analyze a sample must be based on the use of either in-line monitoring or an onsite radiological and chemical analysis facility, and must provide, as needed, quantification of the following:

(i) Those radioisotopes necessary to determine the degree of core damage;

(ii) Hydrogen in the containment atmosphere;

(iii) Total dissolved gases and dissolved hydrogen gas in the reactor coolant;

(iv) Boron in the reactor coolant; and

(v) Chloride in the reactor coolant.

(d) *Leakage Integrity Outside Containment.*

(1) Each boiling and pressurized light-water nuclear power reactor licensee shall implement leak reduction measures so that leakage from systems outside containment (systems that would or could contain highly radioactive fluids during and following a serious transient or accident) is eliminated or minimized to the maximum extent practicable to prevent the release of significant amounts of radioactive material during and following an accident. Consideration shall be given to reductions of potential release paths that could result from design or operator deficiencies.

(2) Each boiling and pressurized light-water nuclear power reactor licensee shall establish and implement a program of preventive maintenance to eliminate or minimize, to the maximum extent practicable, leakage from systems outside containment. This shall include periodic integrated leak tests at intervals not to exceed each refueling cycle as well as the reduction of potential release paths by appropriate operator training.

(e) *Accident Monitoring Instrumentation.* Each boiling and pressurized light-water nuclear power reactor shall have the capability during and following an accident for:

(1) Providing and recording in the control room a continuous indication of:

(i) Containment pressure by January 1, 1981[2];

(ii) Hydrogen concentration in the containment atmosphere by October 1, 1981;

(iii) Containment water level by January 1, 1981[2];

(iv) Containment radiation level by October 1, 1981;

(v) Radioactive noble gas concentrations in the effluents by October 1, 1981; and

(2) Quantifying the concentration of radioiodines and radioactive particulates in the airborne effluents at each anticipated release point by October 1, 1981.

(3) All the instruments and monitoring systems used for accident monitoring shall be designed and qualified (with extended ranges) to perform their function following an accident characterized by the radioactive material release terms described in paragraph (b)(1)(i) of this section.

(f) *Detection of Inadequate Core Cooling.*

(1) Each boiling and pressurized light-water nuclear power reactor licensee shall develop and implement procedures and training to be used by the operators to recognize the existence of inadequate core cooling and low coolant level in the reactor core using available instrumentation.

(2) Each pressurized light-water nuclear power reactor shall be provided with a primary coolant saturation meter (subcooling meter) that provides in the control room a continuous, recorded, on-line indication, of the primary coolant saturation condition.

(3) By January 1, 1982, each boiling and pressurized light-water nuclear power reactor shall be provided with instrumentation such as a reactor vessel water level indicator which supplies to the control room a recorded, unambiguous, direct indication, of inadequate core cooling. The indication must cover the complete range from normal operation to complete core uncovering and give advance warning of the approach of inadequate core cooling.

(4) All instruments used to detect the existence of inadequate core cooling shall be designed and qualified to perform their function following an accident characterized by the radioactive material release terms described in paragraph (b)(1)(i) of this section.

(g) *Training to Mitigate Degraded Core Accidents.* By April 1, 1981, each boiling and pressurized light-water nuclear power reactor licensee shall include in its training program for all operating personnel training to recognize, control and mitigate the consequences of accidents in which the core is severely damaged. The training shall include the use of all available structures, systems and components that can control or mitigate degraded core accidents.

(Secs. 103, 161b., Pub. L. 83–703, 68 Stat. 936, 948; Sec. 201, as amended, Pub. L. 93–438, 88 Stat. 1242 (42 U.S.C. 2133, 2201(b), 5841).)

[2] Or 30 days after the effective date of the rule, whichever is later.

Dated at Washington, D.C., this 26th day of September 1980.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

Commissioner Gilkinsky's Separate Views Regarding Publication of the Proposed Rule—Interim Requirements Related to Hydrogen Control and Certain Degraded Core Considerations

As the summary of this Federal Register Notice states at the outset, "the accident at Three Mile Island, Unit 2, resulted in . . . generation of hydrogen from fuel cladding-water reaction well in excess of the amounts required to be assumed for design purposes by the current Commission regulations." Yet, in approving this Notice, the Commission in effect proposes to retain the outdated regulations. The proposed rule does little more in the area of hydrogen control than to prevent elimination of a protective measure now in place—use of a nitrogen atmosphere in the containment of boiling water reactors to prevent hydrogen buring—where such elimination might be permitted by current regulations. In the case of pressurized water reactors with ice condenser containments and the latest design boiling water reactors the proposed rule does not advance the level of protection at all. It is this aspect of the proposed rule that I disagree with.

The existing rule (10 CFR 50.44) is based on a view that only a small amount of hydrogen can be generated during an accident. Prior to the Three Mile Island accident, reaction of as much as one percent of the reactor fuel's zirconium cladding with water was regarded as extremely unlikely, even in a major loss of coolant accident. Now we know that larger quantities of hydrogen might be generated in lesser accidents—as much as 50 percent of the zirconium cladding in the TMI–2 core is estimated to have reacted with water thereby releasing several hundred kilograms of hydrogen to the containment. In proposing not to require additional hydrogen control measures in ice condenser containments, the staff and Commission have argued that the changes brought about after the Three Mile Island accident have now really made the probability of generating large quantities of hydrogen insignificant. That assumes a rather more complete understanding of the operational characteristics of large power reactors than I believe experience shows. The Commission should have proposed that all powers reactors demonstrate an ability to cope effectively with amounts of hydrogen roughly comparable to that generated at Three Mile Island.

Commissioner Bradford agrees with these comments except for the statement suggesting that the publication of the rule for comment "in effect proposes to retain the outdated regulations."

[FR Doc. 80-30595 Filed 10-1-80; 8:45 am]

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10 CFR Part 50

Domestic Licensing of Production and Utilization Facilities; Consideration of Degraded or Melted Cores in Safety Regulation

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Advance notice of proposed rulemaking.

SUMMARY: The U.S. Nuclear Regulatory Commission is considering amending its regulations to determine to what extent commercial nuclear power plants should be designed to cope with reactor accidents beyond those considered in the current "design basis accident" approach. In particular, this rulemaking would consider the need for nuclear power plant designs to be evaluated over a range of degraded core cooling events with resulting core damage and the need for design improvements to cope with these events.

This advance notice of proposed rulemaking is being issued to invite advice and recommendations on several questions concerning design and operational improvements for dealing with degraded core cooling. Therefore, the preliminary views expressed in this notice may change in light of comments received. In any case, there will be an opportunity later for additional public comment in connection with any proposed rule that may be developed by the Commission.

DATES: The comment period expires December 31, 1980.

ADDRESSES: Interested persons are invited to submit written comments and suggestions to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch. Copies of comments received by the Commission may be examined in the Commission's Public Document Room at 1717 H Street NW., Washington, D.C. Comments may also be delivered to Room 1121, 1717 H Street NW., Washington, D.C., between 8:15 a.m. and 5:00 p.m.

FOR FURTHER INFORMATION CONTACT: M. S. Medeiros, Jr., Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 or telephone (301) 443–5913.

SUPPLEMENTARY INFORMATION:

Historical Background

The Nuclear Regulatory Commission (NRC) is responsible for licensing and regulating nuclear power plants. Before a nuclear power plant can be built at a particular site, a construction permit must be obtained from the NRC. As a major part of the application for a construction permit, the applicant files a Safety Analysis Report. This report presents the design criteria and preliminary design information for the proposed nuclear power plant and provides information on the proposed site. The report also discusses various abnormal conditions and accident situations and describes safety features to be provided to prevent accidents or, if they should occur, to mitigate their effects on public health and safety.

In nuclear power plants, large amounts of radioactive material are generated during fission of nuclear reactor fuel. Although this radioactive material generally remains in the fuel pellets, significant amounts can be released to the reactor coolant during accident conditions. For appreciable amounts of radioactive material to be released from the fuel, it must experience damage from one or more of several possible causes. For example, a hydraulic-mechanical accident at normal fuel temperatures can burst fuel cladding resulting in release of radioactive material normally retained in the gap between the fuel pellets and the fuel clad. A more serious type of accident involving higher fuel temperatures might, in addition to rupturing fuel cladding, cause oxidation of the cladding. This, in turn, would cause hydrogen to be generated and released which would compound the severity of the accident. A still more serious accident might involve very high fuel temperatures and oxidation of a large fraction of the core's zirconium. In this case, not only would large amounts of hydrogen be released to the containment building, but other thermal reactions could result in the release of radioactive material normally held captive in the fuel pellets. Finally, an accident so severe that core melting occurs could release large amounts of radioactive material to the environment if reactor containment integrity were also to be lost.

Based on these considerations, a broad range of nuclear power plant abnormal conditions and accidents with

the potential to cause fuel clad damage and release of radioactive material to the environment has been identified and categorized for analysis. Attempting to prevent abnormal conditions and accidents and mitigating their potential consequences have been the primary objectives of nuclear power plant safety design. The Safety Analysis Report is a key analysis document supporting the adequacy of this aspect of nuclear power plant design.

As discussed in Title 10, Chapter 1, Code of Federal Regulations, § 50.34(a), in the Safety Analysis Report the applicant is required to determine margins of safety for both normal and abnormal operations and to determine "the adequacy of structures, systems, and components provided for prevention of accidents and the mitigation of the consequences of accidents." To assist the applicant in complying with this regulation, the NRC has published Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,[1] which describes the information to be provided in the Safety Analysis Report. In particular, section 15 of Regulatory Guide 1.70 provides guidance to an applicant concerning "design basis assumptions acceptable to the NRC for purposes of determining adequacy of the plant design to meet 10 CFR Part 100 criteria." Regulatory Guide 1.70 explains that these design basis assumptions can, for the most part, be found in regulatory guides that deal with radiological releases and suggests use of Regulatory Guides 1.3 and 1.4, Assumptions Used for Evaluation of the Potential Radiological Consequences of a Loss-of-Coolant Accident.[1] Regulatory Guide 1.70 further states that "This analysis should be referred to as the 'design basis analysis.' " Operating events corresponding to design basis assumptions are termed "design basis accidents," and satisfactory analysis conclusions concerning them allow a judgment that the facility can be operated without undue risk to the health and safety of the public.

It should be noted that these events are analyzed primarily for the purpose of establishing the adequacy of engineered safety features, such features being those structures, systems, and components, designed into a plant to mitigate the consequences of postulated design basis accidents, and which supplement other plant features designed to meet performance specifications for normal operations and anticipated abnormal conditions.

[1] Available from the U.S. Nuclear Regulatory Commission, Washington, DC 20555

In the Safety Analysis Report the applicant is not required, however, to explicitly analyze accidents more severe than the design basis accidents.[2] This approach was based on the assumption that such accidents are of sufficiently low probability that mitigation of their consequences is not necessary for public safety. This low probability was thought to result from the "defense in depth" approach that requires conservative design, multiple physical barriers, quality assurance for design, manufacture and operation, and continued surveillance and testing to prevent such accidents.

Furthermore, in reviewing reactor plant designs using the "design basis accident" approach, the NRC does not review all structures, systems, and components but rather reviews, in varying levels of detail, only those considered "safety grade" by the applicant submitting a Safety Analysis Report. Items considered by the applicant to be outside the scope of design basis accident analyses are generally not considered to be "safety grade" and are not reviewed by the NRC to see whether they will perform as intended or meet various dependability criteria. This method of classification is based on the notion that things credited in the analysis of a design basis event or specified in the regulations are important to safety and thus are "safety grade" while all else is "non-safety grade." Non-safety grade items do not receive continuing regulatory supervision or surveillance to see that they are properly maintained or that their design is not changed in some way that might interact negatively with other systems. Instead, these items simply receive what attention may be dictated by routine industrial codes and by desires to enhance plan availability.

Historically, a further assumption in design review and licensing was that if reactor plant systems can handle large-scale design basis accidents, they can also handle a spectrum of smaller accidents that are regarded as being "within the design envelope."

The accident at Three Mile Island resulted in core damage more severe than that considered in current design basis events [4] and has shown the need to re-examine these historical approaches to analyzing reactor plant design and plant accidents. The October 1979 Report of the President's Commission on the Accident at Three Mile Island [5] recommended that in-depth studies be initiated on the probabilities and consequences (onsite and offsite) of nuclear power plant accidents, including the consequences of core meltdown. This report recommended that these studies include a variety of small-break, loss-of-coolant accidents and multiple-failure accidents, with particular attention to human failures. The report stated that "from these studies may emerge desirable modifications in the design of plants that will help prevent accidents and mitigate their consequences. For example consideration should be given to equipment that would facilitate the controlled safe venting of hydrogen gas from the reactor cooling system," and "consideration should be given to overall gas-tight enclosure of the let-down/make-up system with the option of returning gases to the containment building."

Similarly, the January 1980 report, Three Mile Island, A Report to the Commissioners and to the Public,[6] states, "* * * we have come far beyond the point at which the existing, stylized design basis accident review approach is sufficient. The process is not good enough to pinpoint many important design weaknesses or to address all the relevant design issues. Some important accidents are outside or are not adequately assessed within the 'design envelope'; key systems are not 'safety related'; and integration of human factors into the design is grossly inadequate."

Commission's Intentions

Accordingly, it is the Commission's intent to determine what changes, if any, in reactor plant designs and safety analyses are needed to take into account reactor accidents beyond those considered in the current design basis accident approach. Accidents under consideration include a range of loss-of-core-cooling, core damage, and core-melting events both inside and outside historical design envelopes. Furthermore, the Commission will

[2] There are other design requirements which would presuppose events where significant core damage and release of radioactive material has occurred. For example, radioactive source terms of Technical Information Document TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites,[3] which imply a major reactor accident, are used to judge design adequacy of various engineered safety features and certain other plant systems and components.

[3] Available from National Technical Information Service, U.S. Department of Commerce, Springfield, Virginia 22151.

[4] However, there was no release of fission products from the core more severe than that presumed in 10 CFR Part 100 or TID-14844.

[5] "The Need for Change: The Legacy of TMI," available from the U.S. Government Printing Office, Washington, D.C. 20402.

[6] Copies may be obtained from the GPO Sales Program, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

consider whether to require more coherent consideration of this range of core damage events in the design of both normal operating systems and engineered safety features. Therefore, this advance notice of proposed rulemaking is being published to provide the regulated industry and the public an opportunity to provide advice and recommendations to the Commission on what should be the content of a regulation requiring improvements to cope with degraded core cooling and with accidents not covered adequately by traditional design envelopes. The rulemaking proceeding will address the objectives of such a regulation, the design and operational improvements being considered, the effect on other safety considerations, and the costs of such design improvements compared to expected benefits.

Recognizing the need for prompt action to correct specific deficiencies identified during the Three Mile Island accident and subsequent investigations, the Commission is publishing a proposed interim rule that would require certain interim improvements to better cope with degraded reactor cores. That proposed interim rule should not be viewed as prejudging the final action concerning this advance notice of proposed rulemaking, and comments should be framed accordingly.

In addition to this **Federal Register** Notice, the Commission's Office of Standards Development is making a direct mailing to affected licensees and other known interested persons to ensure that they are aware of this advance notice of proposed rulemaking.

Summary of Features Being Considered for Proposed Rule

The Commission is considering initiating rulemaking that would require that a broader range of accidents of both lesser and greater severity than the design basis accident, including a fully melted core, be considered in plant design, plant operation, and reactor safety analyses.

Specific Considerations

Advice and recommendations on a proposed rule reflecting the foregoing feature and on any other proints considered pertinent are invited from all interested persons. Comment is also invited on the extent to which any additional measures should be backfitted. Comments and supporting reasons are particularly requested on the following questions:

1. If loss of core cooling and resultant core damage occur in a nuclear power plant, there are certain predictable consequences. Can these consequences be mitigated substantially, and the risk of severe public health danger thereby reduced substantially, by practicable design improvements? If not, why not, or, if so, what design improvements can be made and at what estimated cost? How would your recommendations affect other safety considerations?

2. The Three Mile Island accident was terminated after the core was damaged severely but before substantial melting occurred, a condition beyond the current design-basis-accident events considered in the safety analysis. Should the NRC require that events of this type be considered in future safety analyses? If not, why not, or, if so, what criteria would you impose to judge design acceptability?

3. Although the consequences of core-melt accidents have been considered to some extent in assessing nuclear power plant safety, such as in requirements for siting, emergency response plans, and certain engineered safety features, explicit consideration of the capability of current designs and casulty procedures to cope with core-melt accidents has not been a part of safety analysis scrutiny by the NRC. Should core-melt accidents be specifically evaluated in safety analysis reviews, and, if so, to what extend, or, if not, why not?

4. Recognizing that there can never be complete assurance that only analyzed events as delineated in a Safety Analysis Report will occur, what additional analyses, procedures, or design features would propose to mitigate fuel damage accidents in the range from extensive clad perforation without oxidation, through a few percent clad oxidation, through extensive oxidation to full core meltdown? Would you recommend different and perhaps overlapping design features depending on the severity of core damage to be coped with?

5. To what extent should reactor design and reactor safety analysis account for engineered safety features not working at all, not working well, or being defeated by the operator, resulting in severe core damage? What limits should be placed on multiple failure and operator error assumptions made in safety analyses and how should probabilistic risk assessment be used to determine these limits?

6. Should the NRC require construction, at each nuclear reactor plant site, of a new structure for controlled filtering venting of the reactor containment structure? Would you limit the function of such a new structure to filtering particulates, elemental iodine, and inorganic iodine or would you include adsobption bed systems using charcoal or other processes so that organic iodine and noble gases could also be trapped? What quantities and release rates of gases and particulates would you design such a structure to handle and at what removal efficiency and cost? Do the potential reductions in risk expected from such a structure offset potential increases in risk that may materialize from incedents such as inadvertent operation or the concentration of hydrogen in the filtering apparatus?

7. Should the NRC require incorporation into containment design, systems for controlling combustion of hydrogen? Do you favor methods of control that suppress combustion or do you favor controlled burning? If you favor suppression of combustion, what techniques would you recommend and should they vary as a function of the design capability of current containments? If you favor controlled burning, do you recommend open flames, spark plugs, catalytic combustors, or some other means? What percent of zirconium oxidation in the core and at what rate would you design for? Would you respond differently for different reactor or containment types? If so, what differences would you recommend?

8. Would you recommend that all nuclear power plants operate with a nitrogen-enriched containment atmosphere as some BWR plants currently do? Why or why not and, if not, to which types of containment, if any, would you limit required nitrogen enrichment?

9. Should the NRC require incorporation into containment design, a core retention system to mitigate the consequences of core meltdown by, for example, increasing resistance to molten core debris penetration and thereby substantially reducing gas, vapor and aerosol generation to less than that which occurs when core debris is allowed to interact with concrete? Assuming a core retention system is required, do you favor a device that delays melt-through of the containment basemat, or a device that permanently retains core debris within the containment building? If you favor delay of core melt-through, do you recommend refractory materials (such as MgO, ZrO_2) to protect the containment concrete basemat, or do you recommend some other means? If you favor permanent retention of core debris, do you recommend using refractory materials in combination with cooling systems that rely either on natural convective cooling or forced pumping of coolant around the extremities of the

refractory material, or do you recommend some other concept? Would you respond differently for different containment types? If so, what differences would you recommend? How do your recommendations affect other safety considerations?

10. Should the NRC require design changes to account for increased radioactive material that may be transported during an accident by systems normally functioning with much lower levels of radioactivity such as the steam and residual heat removal systems and the containment drainage system?

11. Should the NRC require more extensive operator training, strict literal compliance with new and improved detailed operating procedures, increased reliability of emergency cooling or decay heat removal capability, and expanded control room minimum manning as alternatives or supplements to degraded cooling design improvements?

12. Should the NRC require an alternate, add-on, self-contained decay heat removal system to prevent degradation of the core or to cool a degraded core, in contrast to the previously discussed schemes which are aimed toward mitigating the consequences of degraded core cooling? How would such a decay heat removal system affect other safety considerations?

13. Should the NRC require systems such as the makeup and purification systems to be located in a leak-tight building? Would such a requirement add to or detract from overall plant safety?

14. What design, quality and seismic criteria would you recommend for any additional systems to prevent the potential breeching of containment such as systems for controlled filtered venting, hydrogen combustion control, and core retention mentioned in previous questions? Do you favor evaluating designs of such systems on a realistic basis, as opposed to the conservative method used to evaluate engineered safety features? Do you favor establishing design criteria for such systems that are equally stringent, less stringent, or more stringent than those applied to engineered safety features? Please explain your response in terms of criteria you would recommend, including consideration of redundancy, diversity, testability, inspectability, and structural design limits (including seismic requirements).

15. Can probabilistic analysis be used both as an aid in determining and comparing the adequacy and usefulness of the several features mentioned in previous questions and as an aid in determining the design criteria and reliability requirements for these features? How do you view the utility of quantitative risk analysis in better understanding the safety advantages and disadvantages of the several features mentioned in previous questions?

16. In weighing the costs of design and operational improvements to cope with degraded core cooling against the benefits of their use, what quantitative methods or other guidance would you suggest to facilitate preparation of a useful value-impact assessment? Would you consider useful or appropriate comparisons between nuclear power plant risks and other risks to which people are exposed?

17. What aspects of degraded cooling or melted-core accidents are sufficiently unknown or uncertain as to impede design and analysis of mitigating systems, thus requiring additional research or experimentation?

18. The NRC has under way a separate rulemaking proceeding concerning reactor siting and an emergency planning rule has recently been approved. If you are familiar with these separate activities, how would you modify present and proposed requirements for emergency planning and reactor siting if accidents beyond the present design basis were to be considered in nuclear power plant safety analyses?

Dated at Washington, D.C., this 26th day of September 1980.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

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